Comparative Investigation for Uncertainty Analysis on LOCA of i-SMR

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1. Introduction

Small modular reactors (SMRs) are widely studied and developed around the world owing to their inherent safety, low cost, and site flexibility [1]. Due to relatively low barriers to adopting SMR technology, ensuring robust safety standards is a critical requirement imposed by regulatory authorities. One of the examples of safety could be found in i-SMR, the SMR developed by Korea Hydro & Nuclear Power Co., Ltd. The company has established top tier requirements for the new reactor type to satisfy enhanced safety [2]. i-SMR achieves inherent safety by incorporating a passive containment cooling system (PCCS), passive emergency core cooling system (PECCS), and passive auxiliary feedwater system (PAFS).

However, despite these safety systems, the inherent properties of solid fuel make it imperative to analyze the possibility of a severe accident (SA) [3]. To analyze severe accidents in SMRs, system codes are utilized for accident analysis, providing insights into the safety of SMRs. However, due to the complexity of severe accident phenomena and the limitations of related models, the analysis results inherently involve uncertainties. Therefore, to verify whether the safety of SMRs falls within the target range, it is essential to quantify the uncertainties and assess the sensitivity of the related models.

In this study, a severe accident scenario is postulated, and the behavior of the base case is analyzed. The sensitivity and uncertainty analyses are also conducted on key parameters, including hydrogen mass production, corium mass relocated, and the timing of SA management guideline (SAMG) entry conditions. The uncertainty analysis provides insights into the calculation ranges of these parameters, and comparing these results with sensitivity analysis enhances the reliability of the system code.

2. Methodology

In this study, uncertainty analyses were performed by using the parameters listed in Table I. These parameters were selected based on the critical importance of SA phenomena. The analysis results were derived through the selection of Figures of Merit (FOM) that represent severe accident phenomena and the comparative evaluation of these values. The FOMs are presented in **Table II**.

Table I: Uncertainty parameters for LOCA analysis

Parameters	Explanation
VF_c	Radiative heat transfer coefficient
Exp_blockage	Exponential value of flow blockage
TZr_melt	Zr melting temperature
$TZrO_2$_melt	ZrO_2 melting temperature
TUO_2_melt	UO_2 melting temperature
Tcrit_slump	Slumping start temperature
Eta_slump	Maximum slumping height fraction
T1_oxid	Oxidation model minimum temperature
T2_oxid	Oxidation model maximum temperature
Debris_diam	Molten corium jet behavior debris particle diameter
C_db_qnc	Debris quenching heat transfer coefficient
H_mn_max	Steam maximum heat transfer coefficient

Table II: Figure of merits for LOCA analysis

FOM	Description
Hydrogen generation	Total hydrogen mass
Corium relocated mass	Corium relocation to lower plenum mass
SAMG entrance timing	CET > 923K

Three of the FOMs were selected to evaluate the most important factors of the SA sequence.

2.1 SA Analysis code: CINEMA

CINEMA (Code INtegrated severe accident Evaluation and MAnagement) is a system code developed autonomously in South Korea. CINEMA consists of four main packages which are, CSPACE, SIRIUS, SACAP, and MASTER [9]. Each of the module has different roles such as CSPACE interprets thermal-hydraulic behavior, SIRIUS examines the fission products, and SACAP is utilized to observe the phenomena occurred within containment building. The MASTER package oversees the interaction between these modules. Since i-SMR does not have containment building and has target FOMs limited to thermal-hydraulic phenomena, SACAP and SIRIUS modules are not regarded in this study. The version utilized in this study is CSPACE 2.0.2.356 and MASTER 2.0.2.131.

2.2 Scenario Description

The postulated scenario is loss of coolant accident (LOCA) from the pipe rupture in a modular makeup purification system (MMPS) letdown line with a 2-inch of diameter in break size. The availability assumption for PECCS is EDV all available (2/2), and ERV unavailable (0/2). The figure of i-SMR and pipelines are demonstrated in **Fig. 1**. The overall accident progression of the base case is represented in the **Table III**.

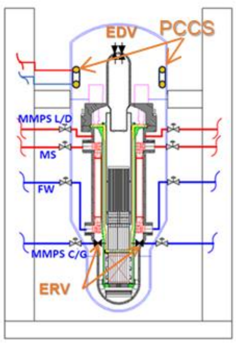

Fig 1. Diagram of i-SMR valves and pipelines

Table III: Accident progression of LOCA analysis

Event	Timing(s)
INCV upper LOCA	0
Rx, RCP, MFWP trip	12.27
Core uncover start timing	18.48
Cladding Oxidation	28311.3
Gap Release	28400.3
SAMG entry timing (CET > 923.15K)	29167.9
Core dry out	38088.1
Relocation of Core material to lower plenum	51002.8

The base case scenario is described in **Fig 2**.

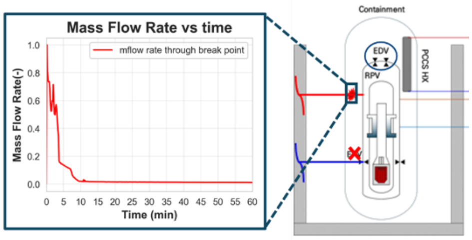

Fig 2. Diagram of i-SMR and break location with mass flow rate through break point

The postulated scenario is demonstrated in **Fig. 2**. As the break occurs, the high-pressure hot coolant gets discharged to the containment vessel. Such action leads to equalization of pressure for reactor pressure vessel (RPV) and containment vessel (CV). The pressure equalization is demonstrated in the **Fig. 3**.

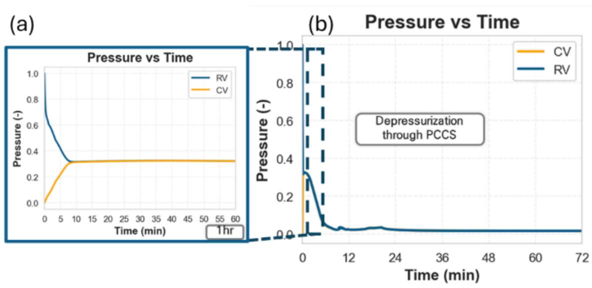

Fig 3. Pressure between RPV and CV
(a) for 1hour (b) for 72 hours analysis

Overall, despite being under extreme condition, the pressure of the CV remains lower than design pressure of CV with the help of heat removal from PCCS. As the scenario begins, the water level of the core decreases due to drop in pressure and saturated temperatures for liquid. The water boils and leaves the core which leads to drop in water level of the core. Such phenomena are demonstrated in **Fig. 4.**

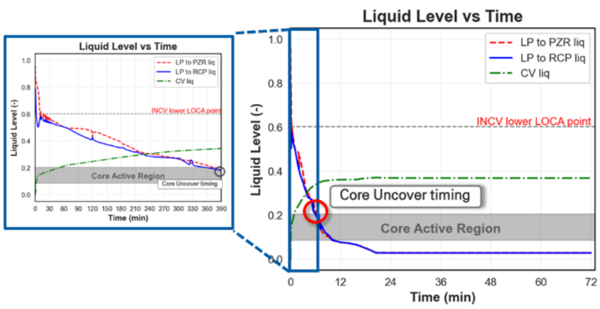

Fig 4. Water level analysis
(a) from the beginning to core uncover timing
(b) for the whole case scenario

As illustrated in **Fig 4**, the water level within the core could not be recovered due to unavailability of ERV. The absence of the coolant in core region lead to rise in the temperature of the core as demonstrated in **Fig 5**.

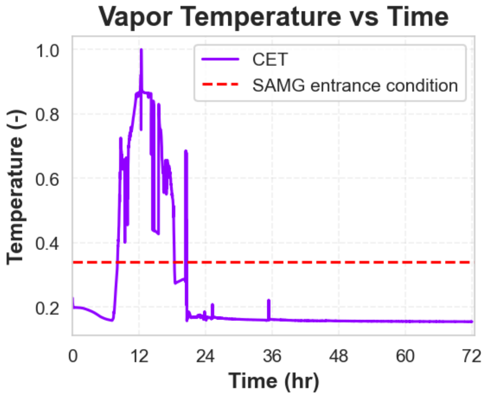

Fig 5. Core exit temperature with SAMG condition

The core temperature rises to over oxidation temperature which leads to the reaction of the cladding with steam. The oxidation between the steam and cladding produce hydrogen as illustrated in **Fig 6**. Owing to the oxidation, the temperature of the core region rise significantly leading to melt in fuel assemblies as demonstrated in **Fig 7**.

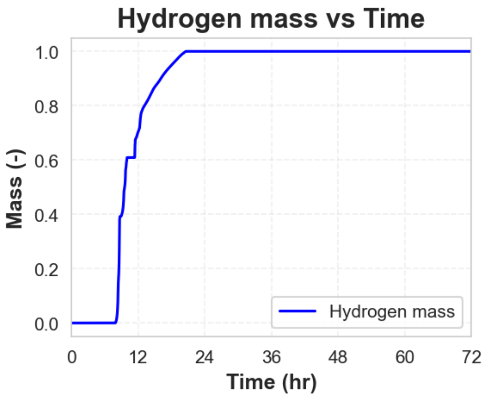

Fig 6. Scaled hydrogen mass production in this scenario

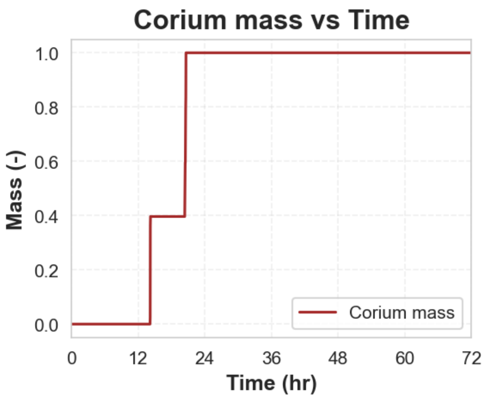

Fig 7. Total corium mass scaled in the scenario

The hydrogen and corium masses produced in the base case scenario are depicted in Figures 6 and 7. Evaluating both the amount and range of mass generated is crucial for assessing safety of the design. However, ensuring the reliability of the system code is essential prior to this evaluation. Thus, in this study, quantification of the uncertainty analyses were conducted to measure the consistency of the results. The calculations were performed by providing random inputs within specified ranges. The resulting ranges of hydrogen and corium mass production, along with the timing for SAMG entrance conditions, are investigated. Quantifying these FOMs demonstrates the behavior of the system code under the postulated accident conditions, thereby enhancing its reliability by confirming consistent results.

2.3 Derivation of the scenario numbers: Wilk's formula

The number of scenarios for uncertainty analysis was calculated based on Wilk's formula. Wilk's formula is a non-parametric statistical method that enables the finding of the appropriate sample size regardless of the underlying probability distribution [4,5]. With this formula, several studies estimated the minimum required sample size and simulated the number of scenarios for analyzing uncertainty [6,7,8]. The formula is written as follows:

$$Wilks' Formula: \sum_{k=0}^{n-p} \binom{n}{k} \alpha^k (1-\alpha)^{n-k} \geq \beta$$

Where n is the desired number of datasets, p is the order of Wilks' formula, α is the cumulative distribution function value, and β is the coverage. According to [2], the 95% of prediction (β) and 95% of coverage (α) give 59 scenarios for 1st order and 93 scenarios for 2nd order. The order of the formula indicates the smallest value among the resulting distribution of the outputs. With the resulting calculation, the 1st and 2nd orders in the Wilks' formula is utilized for uncertainty analysis. Hence 59 and 93 scenarios are sampled and compared after running the simulation of the integrated severe accident analysis code, CINEMA.

2.4. Sensitivity analysis and uncertainty analysis

Uncertainty analysis is conducted due to two of the main reasons: First is, under the condition of severe accident, the complex simultaneous phenomena within the nuclear power plant cannot be exactly estimated [9]. Therefore, by performing uncertainty analysis, the behavior range and quantification of specific results could be obtained. Thereby providing insight in the specific accident scenario that could support the operator even under the harsh condition of the nuclear power plant. Second, the system code inherently brings uncertainty due to the innate interpretation of the SA and different computer capability environment. Therefore, analyzing the result of the different scenario numbers and comparing it with various condition is expected to enhance the reliability of the system code. For uncertainty analysis, the number of target scenarios were selected as 59 and 93 through Wilks' formula 1st order and 2nd order. With the given number of scenarios, the parameters were randomly selected. The result of the analysis is compared with the results of the uncertainty analysis.

3. Result

3.1 Hydrogen mass comparison

The first FOM is the hydrogen generation. The results of sensitivity analysis and uncertainty analysis in 59 case scenarios with 93 case scenarios are demonstrated in **Fig. 7,8**.

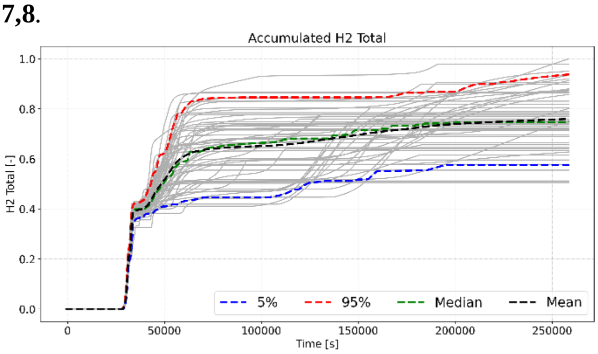

Fig 7. Uncertainty analysis of 59 cases for hydrogen mass production

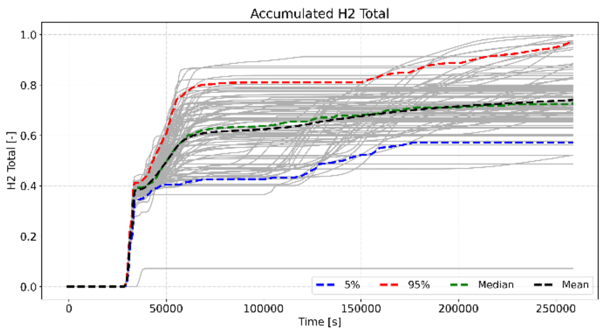

Fig 7. Uncertainty analysis of 93 cases for hydrogen mass production

Table IV: Scaled results of hydrogen mass analysis

Hydrogen mass	Uncertainty Analysis (59)	Uncertainty Analysis (93)
95%	0.899	0.959
Mean	0.728	0.737
Median	0.718	0.715
5%	0.552	0.586

Table IV indicates the uncertainty analyses normalized results of hydrogen mass production in the LOCA. The results showed that the mean values are almost as similar by showing within 1% of the difference. For the range from 95% to 5%, the differences are within 6.25%. Although some of the ranges are not perfectly in accordance, it could be concluded that the resulting values of uncertainty indicate the similar results. Such concordance could enhance the calculation reliability of CINEMA code and i-SMR interpretation, specifically in hydrogen mass calculation.

3.2 Corium mass comparison

The second FOM is the corium mass accumulated on the lower plenum of the core. The results are demonstrated in **Fig. 9, 10**, and Table IV.

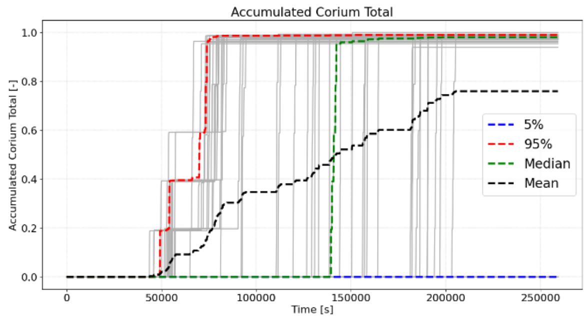

Fig 9. Uncertainty analysis of 59 cases of Corium mass production

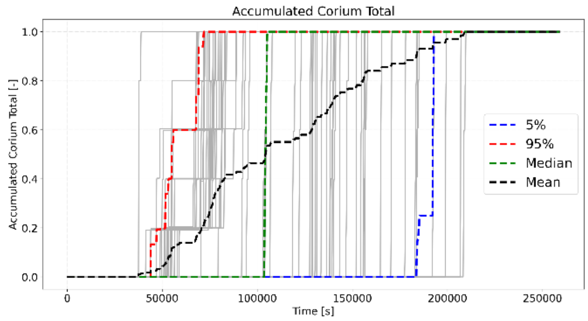

Fig 10. Uncertainty analysis of 93 cases of Corium mass production

Table IV: Scaled results of corium mass analysis

Corium mass	Uncertainty Analysis (59)	Uncertainty Analysis (93)
95%	0.996	0.999
Mean	0.764	0.700
Median	0.986	0.988
5%	0.000	0.000

95%, 5%, and median values are in accordance with each other by having less than 1% difference. Whereas the value for mean show 8% of difference. Such difference is from some of the scenarios not having relocation in uncertainty analysis. The 59 cases and 93 cases of scenarios show 11 and 27 cases of non-relocation event which is why 5% of the total scenarios demonstrate 0.0 for the evaluation. The difference in evaluation comes from the uncertainty of melt progress in RPV. According to sensitivity analysis, TUO_2 melting temperature was the crucial factor for the relocation in corium. When the TUO_2 melting temperature was set high, the temperature within the core could not reach the melting temperature, leading to the absence of relocation. However, in uncertainty analysis, even if the temperature was set high, the relocation still could not be observed. Further investigation on the complex phenomena of relocation mechanisms is expected to be conducted for the future studies.

3.3 SAMG entrance condition comparison

SAMG is applied to conventional large nuclear power plant in the event of SA. The entrance condition is determined by the core exit temperature going over 923K [12]. Although further research is required, SAMG condition could still be applied for i-SMR for having the same nuclear fuel assembly type. In the event of the SA, the timing of the entrance condition is crucial. The sensitivity analysis and two uncertainty analyses demonstrated the following timings for the SAMG entrance condition. In the base case event, from Table III, SAMG entrance timing is analyzed to be 29167.9 second after the initial event.

Table V: Timing for SAMG entrance condition

SAMG Entrance Timing (s)	**Uncertainty Analysis (59)**	**Uncertainty Analysis (93)**
95%	29,443	29,434
Mean	29,182	29,227
5%	28,954	28,938

	SA Entry timing			
	positive		negative	
+/−	UA(59)	UA(93)	UA(59)	UA(93)
PCC	T1_oxid (0.34)	Eta_slump (0.18)	H_mn_max (−0.57)	Tcrit_slump (−0.35)
SRCC	T1_oxid (0.37)	Eta_slump (0.18)	H_mn_max (−0.55)	Tcrit_slump (−0.33)

Fig 11. Correlation coefficient for SAMG entrance timing

Table V indicates that the calculation ranges of the SAMG entry timing from uncertainty analyses show consistent agreement. Specifically, the uncertainty analysis involving 93 case scenarios yields the largest SAMG entry timing range (496 seconds). Interestingly, despite the difference in scenario numbers, both uncertainty analyses (59 and 93 scenarios) exhibit similar ranges (489 seconds and 496 seconds, respectively), suggesting stability in results.

The difference of approximately 8 minutes (~496 seconds) in uncertainty analyses highlights acceptable uncertainty bounds for practical applications. Thus, both analyses collectively provide credible and consistent insights regarding critical parameters impacting the SAMG entrance timing.

4. Conclusions

In this study, uncertainty analyses were conducted using CINEMA to investigate the total hydrogen production, corium relocation mass, and SAMG entrance timing. The comparison of theses FOMs revealed good agreement between uncertainty analyses, demonstrating consistency within similar result ranges. The results highlight the reliability of the CINEMA code and confirm its capability to simulate hydrogen generation and molten corium behavior accurately under severe accident conditions. For future work, it is recommended to extend the analysis to investigate fission product behaviors using the same the same parameters and to further increase the number of uncertainty analysis scenarios to comprehensively evaluate the robustness and calculation capability of CINEMA.

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